NON-PUBLIC?: N
ACCESSION #: 8802230093

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 of 8

DOCKET NUMBER: 05000410

TITLE: Reactor Scram due to a Loss of Feedwater Flow Caused by Personnel Error
EVENT DATE: 01/20/88 LER #: 88-001-00 REPORT DATE: 2/17/88

OPERATING MODE: 1 POWER LEVEL: 041

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On January 20, 1988 at 0944 hours with the reactor operating at approximately 41% power, Nine Mile Point Unit 2 experienced a scram due to an actual low (Lev
l 3) water level condition. The low water level was caused by a loss of feedwater flow to the reactor. An operator while placing a markup (tag out), valved an instrument air system prefilter out of service without ensuring that a redundant prefilter was in service. This isolated the air compressors from the remainder of the system. Instrument air pressure throughout the plant decayed to the point of causing the minimum flow valves for the condensate, condensate booster, and feedwater pumps to fail open. As a result, reduced feedwater flow to the reactor caused reactor water level to rapidly decrease to the Level 3 scram setpoint (159.3 inches). Both the High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) systems actuated on low level instrumentation which restored normal water level in the vessel. The Main Turbine-Generator also tripped due to RCIC injection.

Design and personnel errors were encountered during the event. Corrective actions have been implemented and/or incorporated to minimize the potential of a recurrence.

(End of Abstract)

I. DESCRIPTION OF EVENT

On January 20, 1988 at 0944 hours with the reactor operating at approximately 41% power in the natural circulation mode per startup and test procedures, Nine Mile Point Unit 2 (NMP2) experienced a scram due to an actual low (Level 3) water level condition. The low water level was caused by a loss of feedwater flow to the reactor.

A Niagara Mohawk operator, while placing a markup, valved Instrument Air System (IAS) prefilter 2A out of service without ensuring that the redundant prefilter 2B was in service. Instrument air pressure throughout the plant began to decay due to this valving error. The licensed operators first noticed the "CRD Scram Valve Pilot Air Header Pressure Low" annunciator and immediately began to investigate. As air pressure decayed, the air operated minimum flow valves for the condensate, condensate booster, and feedwater pumps failed open (Figure 1). As a result, feedwater flow recirculated back to the condenser subsequently reducing feedwater flow to the vessel.

Operating feedwater pumps (P1B & P1C) and condensate booster pump (P2A) tripped on low suction pressure. Condensate pumps P1A, P1B and P1C and condensate booster pump P2B remained running throughout the event. An Operator placed the master feedwater controller in manual in an attempt to restore water level. He soon realized that the feedwater valves (LV10B & C) were fully open and that no flow was going to the reactor. The reactor scrammed on Low (Level 3) water level, 159.3 inches. Reactor level continued to decrease until level reached 112 inches. At which time, the High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling system actuated and restored level.

Instrument air was returned to normal approximately one minute after the scram.

HPCS, the HPCS Diesel Generator, and RCIC systems initiated on a low-low water level signal (Level 2), 108.8 inches. The Main Turbine-Generator tripped due to RCIC injection. Reactor water level was restored and HPCS injection was terminated, by the operators, when level reached 195 inches. RCIC secured automatically at Level 8 (202.3 inches), as designed. The reactor pressure was reduced during this time, due to the injection of cool water from the HPCS and RCIC systems, and the relatively small amount of decay heat in the new core.

Approximately 5 minutes after the scram, the feedwater minimum flow valves closed from a normal control circuit signal. Feedwater system pressure increased to the point of exceeding reactor pressure due to the operating condensate and condensate booster pumps. With the master feedwater controller in manual and the level control valves fully open, feedwater flow resumed to the vessel. Operators later noticed that reactor water level was starting to increase without HPCS or RCIC injecting. An operator manipulated the Manual/Auto feedwater controllers to the full closed position in an attempt to prevent excessive overfilling of the reactor. The feedwater regulation valves (FWS-LV10B, C) locked up at approximately 80% open, when a loss of control signal was sensed by the valve hydraulic actuator control card. This caused the vessel to overfill and flood the main steam lines. The operators determined that feedwater was still injecting into the vessel and closed the feedwater containment isolation valves (MOV21A & B). This terminated the feedwater injection and water level peaked at 333 inches.

The main steam lines were flooded up to the turbine stop valves. Normal steam line drains were utilized to drain the excess water from the steam lines. A turbine bypass valve was later opened manually to control reactor pressure and to initiate a plant cooldown.

The remainder of the scram recovery was routine.

II. CAUSE OF EVENT

A root cause analysis for this event has been completed per Site Supervisory Procedure S-SUP-1, "Root Cause Evaluation Program". The root cause has been determined to be personnel error. The operator visually verified that the isolation valves for prefilter 2B were open by observing the length of the valve stems. However, he should have physically verified the valves' position open. These valves were in fact closed. Thus, when the operator isolated prefilter 2A for maintenance, the instrument air compressors were isolated from the rest of the system.

The root cause of the feedwater regulation valves locking up was design deficiency. An existing ground loop in the control circuitry biased the setpoint upward for the loss of control signal, causing lock-up of the feedwater regulation valves.

III. ANALYSIS OF EVENT

The isolation of the air compressors posed no adverse safety consequences to plant personnel or public safety as a result of this event. This portion of the IAS is not required to affect or support the safe shutdown of the reactor or to perform any safety-related functions associated with its operation. The loss of instrument air caused the minimum flow valves for the condensate, condensate booster, and feedwater pumps to fail open allowing flow to recirculate back to the condenser as designed (FSAR, Section 15.0.5, "Loss of Instrument Air").

The reactor scram which occurred as a result of the Level 3 trip was a conservative action. This poses no adverse safety consequences at any power level. This action did not in any way adversely affect any other safety system nor the operators ability to achieve safe shutdown.

Water level never fell below 112 inches during the transient. HPCS and RCIC initiated well above (conservative direction) the 108.8 inch Technical Specification (TS) required setpoint. Because water level never reached 108.8, some Level 2 system actuations and isolations did not occur. An investigation was performed to ensure the setpoints of all Level 2 instruments were correct. This investigation found that the guide value setpoint for Primary Containment Isolation Actuation Level switches had been calculated incorrectly. Details can be found in LER 88-04.

The automatic actuation of RCIC and HPCS with coolant injection was a conservative response with minimal plant impact and no resultant impact on public safety. The systems operated as designed and restored reactor water level to Level 8, 202.3 inches.

In accordance with the requirements of TS Sections 3.5.1(f) and 6.9.2 (Emergency Core Cooling System (ECCS) Injections, the following special data is provided:

For the HPCS nozzle,

- Total accumulated initiation cycles to date = 1

- Current usage factor value remains well below 0.70

The increasing reactor water level which resulted in the flooding of the main steam lines prompted the following concerns:

- Thermal transients across the main steam pipe and reactor main steam nozzle walls caused by assumed colder water flowing into the hot main steam line.

- The weight of water in the main steam lines.

- Dynamic transient loads caused by water flowing into the main steam lines.

- External loads/moments imposed on the reactor main steam nozzles.

Evaluation of the local stress due to the thermal transient across the pipe nozzle walls conservatively used the temperature of the feedwater as the fluid temperature entering the main steam pipe. This was conservative since it did not account for thermal mixing within the Reactor Pressure Vessel (RPV) due to the natural circulation of reactor water, the effect of core decay heat, or the effect of stored thermal energy in the reactor coolant pressure boundary metal piping, structures, and the reactor core. The results of the evaluation of the thermal transient indicated that all stresses satisfied design code allowables for the piping from the vessel to the main turbine stop valves and drywell penetrations. Pipe break exclusion criteria was also satisfied for this event.

Final evaluations to establish the actual usage factor for plant life will use fluid temperatures calculated by a Contractor Engineering group for this event.

Evaluation of the weight of water in the main steam piping indicated that the design pipe stress analysis included load cases considering water in the main steam lines, therefore, no further review was required. Pipe supports were designed for maximum design loading considering either water or steam. Snubbers and spring hangers were sized to account for the maximum movement due to the combination of plant thermal, dynamic, and deadweight (water or steam) conditions.

Review of linear potentiometer displacement data in the main steam piping indicated that there were no vibratory responses during the flooding event. From this it can be concluded that hydraulic transient affects were insignificant.

The only external loads/moments imposed on the main steam piping, components, and reactor nozzles was the weight of water and thermal expansion of the system. Since no seismic event occurred during the scram and subsequent plant shutdown, the applied loads were enveloped by the design loads. Therefore, no further evaluation was required.

In conclusion, we have conservatively determined that the effects of thermal transients across the pipe and reactor main steam nozzle walls, the weight of

water, and hydraulic dynamic transients, are within applicable code stress allowables for the piping and penetrations and within design pipe support loadings. Break exclusion criteria is satisfied where applicable and RPV no
zle loadings due to this event are enveloped by existing design loads.

In summary, while the main steam line flooding and the loss of instrument air was an undesirable transient, effects of the transient were within plant design margins.

IV. CORRECTIVE ACTIONS

1. The individuals involved with this event participated in the LER investigation and follow-up report.

2. The operator involved in the valving error has been subjected to disciplinary action for inattentiveness to duty.

3. This event has been incorporated into the Operations Department Lessons Learned Program.

4. Administrative Procedure AP-3.3.1 has been revised regarding valve position verification policy.

5. The valve position verification policy will be incorporated into the operators requalification training program via TMR 02-88-18.

6. Modification PN2Y87MX238 - This modification incorporated a design change to the control circuitry of the manual/automatic controllers of the feedwater regulation valves. This change addresses the problem of inadvertent feedwater control valve lockup.

7. Operating Procedure No. N2-OP-19, "Instrument and Service Air System" has been revised to provide additional direction in an event of loss of instrument air.

8. Operating Procedure No. N2-OP-101C, "Plant Shutdown" has been revised. A caution statement has been added to alert the operators to continuously monitor reactor water level if the feedwater regulation valves are in manual.

9. A review of the Nuclear Steam Supply Shutoff System (NSSSS) level instrumentation revealed that four level switches were found to have improperly calculated guide values. The Instrument and Control

(I&C) Department immediately recalibrated these instruments and incorporated procedure changes as necessary. Additional information is contained in LER 88-04.

10. The guide value calculations for all other level, presure, and temperature instrument channels which have TS required trip setpoints have been reviewed by I&C to assure accuracy. Additional information is provided in LER 88-04.

V. ADDITIONAL INFORMATION

Identification of Components Referred to in this LER

IEEE 803 IEEE 805
Component EIIS Funct System ID

Instrument Air System (IAS) N/A LD
High Pressure Core Spray System (HPCS) N/A BG
Reactor Core Isolation
Cooling System (RCIC) N/A BN
Main Steam System (MSS) N/A SB
Reactor Pressure Vessel RPV N/A
Reactor Water Cleanup System (RWCU) N/A CE
Instrument Air Prefilter FLT LD
Feedwater Pump P SK
Condensate Booster Pump P SG
Turbine Stop Valves ISV TG
Turbine Bypass Valve ISV TG
Valve ISV LD
Annunciator ANN IB
Manual/Auto Feedwater Controller FCO JB

There has been one previous similar event where reactor water entered the MSS lines, LER 86-20.

FIGURE OMITTED - NOT KEYABLE (DRAWING)

ATTACHMENT # 1 TO ANO # 8802230093 PAGE: 1 of 1

NMP29854

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NIAGARA MOHAWK
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THOMAS E. LEMPGES
VICE PRESIDENT--NUCLEAR GENERATION

February 17, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 88-01

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 88-01 Is being submitted in accordance with 10 CFR 50.73(a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR50.72 report for this event was made at 1030 hours on January 20, 1988.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,
/s/ Thomas E. Lempges
Thomas E. Lempges
Vice President
Nuclear Generation

TEL/SCN/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

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